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March 22, 1977

Docket No.: 50-261

Carolina Power & Light Company
ATTN: Mr. J. A. Jones
Senior Vice President
336 Fayetteville Street
Raleigh, North Carolina 27602

Gentlemen:

The Commission has issued the enclosed Amendment No. 27 to Facility Operating License No. DPR-23 for the H. B. Robinson Steam Electric Plant Unit No. 2. The amendment consists of changes to the Technical Specifications in response to your application dated November 15, 1976, and staff discussions.

This amendment incorporates into the Technical Specifications requirements and restrictions to be applied to the Spent Fuel Cask Handling Crane.

Copies of the Safety Evaluation and the Federal Register Notice are also enclosed.

Sincerely,

Original signed by

Robert W. Reid, Chief
Operating Reactors Branch #4
Division of Operating Reactors

Enclosures:
1. Amendment No. 27
2. Safety Evaluation
3. Federal Register Notice

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DATE ▶	2/11/77	2/14/77	2/23/77	3/22/77	2/17/77	2/17/77

Carolina Power & Light Company

cc w/enclosure(s):
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Mr. McCuen Morrell, Chairman
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cc w/enclosures and incoming
dtd.: 11/15/76
Office of Intergovernmental Relations
116 West Jones Street
Raleigh, North Carolina 27603

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

CAROLINA POWER & LIGHT COMPANY

DOCKET NO. 50-261

H. B. ROBINSON STEAM ELECTRIC PLANT UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 27
License No. DPR-23

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Carolina Power & Light Company (the licensee) dated November 15, 1976, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 3.B. of Facility Operating License No. DPR-23 is hereby amended to read as follows:

   B. Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 27 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert W. Reid

Robert W. Reid, Chief
Operating Reactors Branch #4
Division of Operating Reactors

Attachment:
Changes to the Technical Specifications

Date of Issuance: March 22, 1977

ATTACHMENT TO LICENSE AMENDMENT NO. 27

FACILITY OPERATING LICENSE NO. DPR-23

DOCKET NO. 50-261

Revise Appendix A Technical Specifications as follows:

Remove Pages	Insert Pages
3.8-3 - 3.8-5	3.8-3 - 3.8-5

The changed area on the revised page is indicated by a marginal line.

b. The results of laboratory carbon sample analysis from the Spent Fuel Building filter system carbon and the Containment Purge filter system carbon shall show ≥ 90 percent radioactive methyl iodide removal at a velocity within 20 percent of the filter system design, 0.05 to 0.15 mg/m^3 inlet methyl iodide concentration, ≥ 70 percent R.H. and ≥ 125°F.

c. All filter system fans shall be shown to operate within ± 10% of design flow.

d. During fuel handling operations, the relative humidity (R.H.) of the air processed by the refueling filter systems shall be ≤ 70 percent.

e. From and after the date that the Spent Fuel Building filter system is made or found to be inoperable for any reason, fuel handling operations in the Spent Fuel Building shall be terminated immediately.

3.8.3 During the discharge of a full core into the spent fuel pit, the temperature of the spent fuel pool water shall be maintained at or below 150°F. The spent fuel pool water temperature shall be monitored once each shift when the temperature is at or below 125°F. If the temperature exceeds 125°F, it shall be monitored hourly. If the pool temperature reaches 150° F, fuel assemblies will be transferred back to the containment to reduce the pool temperature below 150°F.

3.8.4 <u>Spent Fuel Cask Handling Crane</u>

The following restrictions and requirements shall be applied to the Spent Fuel Cask Handling Crane:

a. Use of the Spent Fuel Cask Handling Crane for lifting operations shall be permitted only when the ambient outside air temperature is greater than 33°F. If the temperature falls below this limit, lifting operations shall be suspended, with the load placed in a safe configuration, until the temperature increases above the limit.

b. Limit switches provided to limit travel of the bridge, trolley, and hoist shall be tested every six months when the crane is not in service, and shall be tested prior to each period of service and on a monthly basis while the crane is in service.

c. Crane ropes shall be inspected in accordance with ANSI B30.2.0-1967 every six months when the crane is not inservice, and shall be inspected prior to each period of service and on a monthly basis while the crane is in service. A crane rope shall be replaced if any of the replacement criteria given in ANSI B30.2.0-1967 are met.

Basis

The equipment and general procedures to be utilized during refueling are discussed in the Final Facility Description and Safety Analysis Report. Detailed instructions, the above specified precautions, and the design of the fuel handling equipment incorporating built-in interlocks and safety features, provide assurance that no incident could occur during the refueling operations that would result in a hazard to public health and safety[1]. Whenever changes are not being made in core geometry one flux monitor is sufficient. This permits maintenance of the instrumentation. Continuous monitoring of radiation levels and neutron flux provides immediate indication of an unsafe condition. The residual heat pump is used to maintain a uniform boron concentration.

The boron concentration of 1950 ppm will keep the core subcritical even if all control rods were withdrawn from the core. During refueling, the reactor refueling cavity is filled with approximately 285,000 gallons of borated water. The boron concentration of this water at 1950 ppm boron is sufficient to maintain the reactor subcritical by at least 10% $\Delta k/k$ in the refueling condition with all rods inserted, and will also maintain the core subcritical even if no control rods were inserted into the reactor[2]. Weekly checks of refueling water storage tank boron concentration ensure the proper shutdown margin[3]. Direct communications allow the control room operator to inform the manipulator operator of any impending unsafe condition detected from the control board indicators during fuel movement.

In addition to the above safety features, interlocks are utilized during refueling to ensure safe handling. An excess weight interlock is provided on the lifting hoist to prevent movement of more than one fuel assembly at a time. The spent fuel transfer mechanism can accommodate only one fuel assembly at a time.

The restriction of not moving fuel in the reactor for a period of 100 hours after shutdown reduces the consequences of a fuel handling accident by providing for decay of short-lived fission products and the reduction of fission gas inventory in any potentially failed fuel. Fuel handling accidents in containment and the Spent Fuel Building have been evaluated by postulating that the failure of all fuel rods in one assembly occurs 100 hours after shutdown[4]. During movement of irradiated fuel assemblies in the spent fuel pool, ventilation exhaust is diverted through HEPA and charcoal filters. During movement of irradiated fuel assemblies in containment, the purge system will be either operable, with exhaust flow passing through HEPA and charcoal filters, or containment isolated.

High-efficiency particulate air (HEPA) filters are installed before the charcoal adsorbers to prevent clogging of the iodine adsorbers for all refueling filter systems. The charcoal adsorbers are installed to reduce the potential release of radioiodine to the environment. The in-place test results should indicate a system leak tightness of less than 1 percent bypass leakage for the charcoal adsorbers and a HEPA efficiency of at least 99 percent removal of DOP particulates. The laboratory carbon sample test results should indicate a radioactive methyl iodide removal efficiency of at least 90 percent on the Spent Fuel Building filter system carbon and the Containment Purge filter system carbon for expected accident conditions. If the efficiencies of the HEPA filters and charcoal adsorbers are as specified, the resulting doses will be less than the 10 CFR Part 100 guidelines for the accidents analyzed. Operation of the fans significantly different from the design flow will change the removal efficiency of the HEPA filters and charcoal adsorbers.

The relative humidity (R.H.) of the air processed by the refueling filter systems should be less than the R.H. used during the testing of the charcoal adsorbers in order to assure that the adsorbers will perform under accident conditions as predicted by the test results. Heaters have been installed upstream of the Spent Fuel Building filters to assure an R.H. of less than 70 percent for the air processed by the Spent Fuel Building filter system. If the R.H. in the Containment atmosphere exceeds 70 percent, operation of the Containment Purge system will be terminated until this specification can be met. If the Spent Fuel Building filter system is found to be inoperable, all fuel handling and fuel movement operations in the Spent Fuel Building will be terminated until the system is made operable.

The temperature limit specified for the fuel cask handling crane is based on the recorded ambient temperature at the time of the 125% load test. The limit is imposed to assure adequate toughness properties of the crane structural materials.

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(1) FSAR - Section 9.5.2

(2) FSAR - Table 3.2.1-1

(3) FSAR - Section 9.5.1

(4) Letters--CP&L to AEC: September 27, 1972; January 23, 1973; and February 9, 1973.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 27 TO FACILITY OPERATING LICENSE NO. DPR-23

CAROLINA POWER AND LIGHT COMPANY

H. B. ROBINSON STEAM ELECTRIC PLANT UNIT NO. 2

DOCKET NO. 50-261

Background

By letter dated March 6, 1974, we requested Carolina Power and Light Company (the licensee) to provide us with analyses and other relevant information needed to determine possible damage at H. B. Robinson Unit No. 2 (Robinson-2) in the event of a spent fuel cask drop caused by a crane system failure and whether design or procedural modifications would be appropriate to reduce the probability of occurrence. Additional information on this subject was requested by our letters of September 25, 1974 and October 24, 1975. Information in response to these requests was supplied by the licensee in his letters of May 14, October 17 and December 26, 1974, April 15, and July 18, 1975, and May 14, 1976. In response to our letter of September 16, 1976, the licensee, by letter dated November 15, 1976, requested changes to the Robinson-2 technical specifications in conformance with procedural commitments made in his earlier submittals. These changes related to the permissible minimum ambient temperature for crane operation, surveillance requirements for crane limit switches and inspection and replacement requirements for crane ropes.

Discussion

The overhead crane handling system for Robinson-2 consists of an overhead, bridge-type crane, spent fuel cask lifting devices, and controls. The overhead crane handling system is used during plant operation for lifting and transporting the spent fuel shipping cask between the spent fuel pool and the cask decontamination/shipping areas. The overhead crane is located outdoors and has a main hoist rated at 125 tons. Removable roof panels provide access to the operating areas. The overhead crane handling system has been designed to minimize the potential of a spent fuel cask drop accident which could result in release of radioactive materials by (1) replacing the trolley on the overhead crane with a new trolley designed to single failure criteria pursuant to NRC Branch Technical Position APCSB 9-1 and (2) restricting the path of travel of the crane and spent fuel cask so that the cask passes over the minimum amount of safety related equipment.

Evaluation

The overhead crane has redundancy in the areas of brakes, gear trains, reeving system, load attaching points, and cask lifting devices, as well as crane control components and systems which are designed fail safe. Based on our review of data provided by the licensee through July 18, 1975, we conclude that the integrated design of crane, controls, and cask lifting devices meet the intent of Branch Technical Position APCSB 9-1 regarding single failure criteria except in the specific areas of seismically induced loadings, maintenance of head and load block alignment following a wire rope failure, provisions for detecting bridge and/or trolley overtravel, provisions for preventing crane damage in the event of load hangup, provisions for preventing two-blocking, and physical restraints to prevent the main hoist drum from dropping as a result of end support failure.

The licensee's response to these concerns were contained in his letter of May 14, 1976. We have evaluated these responses as follows:

With regard to head and load block alignment, the licensee has provided quantitative data on the magnitude of seismically induced pendulum motion of the load. Based on these data we conclude that the effect of these added motions on crane loading are negligible.

With regard to head and load block alignment, the licensee has provided tabular data which gives the maximum amount the spent fuel cask will shift in any direction should one of the dual wire ropes break during lifting. These data cover both the highest and lowest positions of the cask during handling. Based on these data we conclude that the maximum load shift at all elevations will not produce load instability. Accordingly, we find the provisions for maintenance of head and load block alignment acceptable.

As regards prevention of bridge and/or trolley overtravel, the licensee has provided a detailed description of the spent fuel cask crane control system, including normal and restricted path modes of operation. The control system is designed to prevent the spent fuel cask crane from passing over irradiated fuel or operating in an area where load hangup could occur while handling the spent fuel cask. The control system includes limit switches to stop all crane motion should the restricted path boundary be breached, as well as alarms to indicate malfunction of the path control limit switches. We conclude that the control system, in conjunction with administrative controls governing restricted path operation, is adequate to prevent critical loads from passing over irradiated fuel or from being hung up during handling and is, therefore, acceptable.

With regard to preventing damage to the crane (as a result of load hangups) by reducing main hoist motor torque, the licensee states that limiting of main hoist motor torque is not required because the established restricted path for critical load handling is void of any physical obstructions on which a critical load could hangup. Physical inspection of the Robinson-2 facility by NRC personnel has confirmed this lack of obstructions. Accordingly, we conclude that the present main hoist motor torque is acceptable.

With regard to providing additional protection against crane two-blocking, the licensee has agreed to install a power limit switch in the main hoist motor power circuit as a further means of preventing two-blocking of the crane under all circumstances. We conclude that this addition satisfies the single failure criterion with respect to crane two-blocking, and is therefore acceptable.

Regarding support of the main hoist drum in the event of a single failure, the licensee has provided both narrative and sketches to show that physical restraints will prevent the main hoist drum from dropping more than 1/8" in the event of support or bearing failures at one end of the drum shaft. The sketches also show the misalignment in the gear trains as a result of such an occurrence is well within the gear tolerance and will not seriously impair the ability to lower the critical load to a safe, neutral position during such emergency conditions. The licensee also states that the added friction of the drum on the restraints will not cause excessive loading on the gear train during emergency lowering. Based on our review of the above, we conclude the main hoist drum meets the single failure criteria intent and is, therefore, acceptable.

Technical Specifications

The crane reeving system which was designed and constructed in accordance with established crane industry standards prior to development of the NRC Branch Technical Position, does not strictly meet the NRC staff recommended criteria for wire rope safety factors and fleet angles. The purpose of these criteria is to ensure a design which minimizes wire rope stress and thereby provides maximum assurance of crane safety under all operating and maintenance conditions. Because the crane reeving system does not meet these recommended criteria there is a possibility of an accelerated rate of wear of the wire rope. Accordingly, to compensate in these design areas, the licensee, by letter dated November 15, 1976, has proposed technical specifications for wire rope inspection and replacement, the purpose of which is to ensure that the entire length of the wire rope will be maintained as close as practicable to original design safety factors at all times. The inspection/replacement program defined by the proposed technical specifications provides a level of protection equivalent to the methods suggested in our wire rope safety and crane fleet angle criteria and will assure that accelerated wire rope wear will be detected before crane use and satisfies our concerns, and we conclude the crane reeving system is acceptable.

Because the crane was designed and constructed prior to development of the NRC Branch Technical Position, it also is not capable of fully meeting the NRC staff recommended criteria for operating temperatures relative to the nil-ductility transition temperature (NDTT) of the crane material. The crane, however, is designed for outdoor service in accordance with established crane industry standards which specify a minimum operating temperature of 0°F. In addition, the licensee by letter dated July 18, 1975, committed to a minimum operating temperature not less than the temperature recorded at the time of the 125% load test. This commitment was reaffirmed by the licensee by letter dated November 15, 1976, which proposed a technical specification limit on the crane minimum operating temperature of 33°F (stated by the licensee to be the temperature recorded at the time of the 125% load test). We have determined that certain modifications in the proposed technical specifications were required. These have been discussed with and concurred in by the licensee. We therefore conclude that these proposed limits on crane minimum operating temperature (as revised), in combination with the performance of the 125% load test at this minimum temperature and past crane industry experience provide a level of protection equivalent to that suggested in our operating environment criteria and are acceptable.

By letter dated September 16, 1976, we requested that the licensee propose technical specifications setting forth surveillance requirements for the limit switches provided to limit travel of the crane bridge, trolley and hoist. The licensee's proposed surveillance requirements were included in his letter of November 15, 1976. We have reviewed the proposed surveillance requirements submitted by the licensee and find them acceptable.

Based on our review of the crane design, the Technical Specifications, and the additional information provided by the licensee, we find that the cask drop accident has been adequately resolved. Thus, we conclude that fuel cask handling, as proposed with the fuel cask handling crane under the surveillance and operating requirements of the proposed Technical Specifications, is acceptable.

Environmental Considerations

We have determined that the amendment does not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. Having made this determination, we have further concluded that the amendment involves an action which is insignificant from the standpoint of environmental impact and, pursuant to 10 CFR §51.5(d)(4), that an environmental impact statement, or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of this amendment.

Conclusion

We have concluded, based on the considerations discussed above, that: (1) because the amendment does not involve a significant increase in the probability or consequences of accidents previously considered and does not involve a significant decrease in a safety margin, the amendment does not involve a significant hazards consideration, (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (3) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Dated: March 22, 1977

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-261

CAROLINA POWER & LIGHT COMPANY

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 27 to Facility Operating License No. DPR-23, issued to Carolina Power & Light Company (the licensee), which revised Technical Specifications for operation of the H. B. Robinson Steam Electric Plant Unit No. 2 (the facility) located in Darlington County, Hartsville, South Carolina. The amendment is effective as of its date of issuance.

The amendment incorporates into the Technical Specifications requirements and restrictions to be applied to the Spent Fuel Cask Handling Crane.

The application for the amendment complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendment. Prior public notice of this amendment was not required since the amendment does not involve a significant hazards consideration.

The Commission has determined that the issuance of this amendment will not result in any significant environmental impact and that pursuant to 10 CFR §51.5(d)(4) an environmental impact statement, or negative declaration and environmental impact appraisal need not be prepared in connection with issuance of this amendment.

For further details with respect to this action, see (1) the application for amendment dated November 15, 1976, (2) Amendment No. 27 to License No. DPR-23, and (3) the Commission's related Safety Evaluation. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N. W., Washington, D. C. and at the Hartsville Memorial Library, Home and Fifth Avenues, Hartsville, South Carolina.

A copy of items (2) and (3) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, Attention: Director, Division of Operating Reactors.

Dated at Bethesda, Maryland, this 22nd day of March 1977.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert W. Reid

Robert W. Reid, Chief
Operating Reactors Branch #4
Division of Operating Reactors